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CANDU CONTAINMENT SYSTEMS - A REGULATORY PERSPECTIVE

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ABSTRACT

Current regulatory requirements for containment systems have evolved over the years. This evolution has been affected by research information, design decisions and operating experience.

RÉSUMÉ

Les exigences réglementaires actuelles concernant les systèmes de confinement ont évolué au cours des années. L'information provenant de la recherche, les décisions des concepteurs et l'expérience acquise au cours de l'exploitation ont eu une influence sur cette évolution.

1. INTRODUCTION

Power reactor technology is unique in many aspects. Foremost among these aspects is that from the outset of the nuclear industry in the Western world, the need for protection against very improbable accidents was accepted almost without question. Massive containment structures are one outcome of the simple concept that any uncontrolled release of energy and fission products should be contained. That simple concept has given rise, however, to a wide variety of complex designs. While the designs have been influenced by economic and safety considerations, this paper will address only the safety question and our collective success in addressing that question.

2. BACKGROUND TO CURRENT REGULATORY REQUIREMENTS

2.1 Impact of ECCS Effectiveness

The current regulatory requirements for containment systems are documented in a regulatory guide (Ref. 1). That document was prepared largely by listing the basic requirements which had evolved over the previous two decades. Underlying all the requirements are principles which emphasize effectiveness, reliability and testability.

The requirements for effectiveness have come directly from the fundamental single and dual failure licensing requirements of the Atomic Energy Control Board (Ref. 2). These licensing requirements stipulate that, for any serious process failure, the proper operation of all the special safety systems should limit doses to people outside the plant boundary to less than the identified reference dose limit. Also, for any serious process failure combined with failure of any one special safety system the remaining safety systems are required to limit doses to an identified (but greater) reference dose limit. Until about a decade ago, designers considered that for any loss-of-coolant accident (LOCA) the emergency core cooling system (ECCS) would prevent significant fuel failure*. With such an understanding of ECCS effectiveness it is evident that a LOCA, by itself, was not seen to pose a significant challenge to the containment in preventing the release of fission products - there would be no significant quantities of fission products free within containment (designers said). Similarly, dual failures which consisted of a LOCA combined with impairment of containment (e.g. failure of ventilation dampers to close) were considered, by inspection, to meet the dual failure reference dose limits since no significant quantities of fission products were expected to be free within containment. The ability of containment to prevent the release of fission products was set only by the dual failures which involved a LOCA or loss of regulation accident combined with complete failure of the shutdown system**, or a LOCA combined with failure of ECCS. Clearly, these dual failures pose a tremendous challenge to a containment system and very effective systems were necessary in order to meet the dual failure reference dose limits.

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* Exceptions to this statement were end fitting failures where fuel would be ejected from the core and in the case of one reactor a very specific size of break in the heat transport system which would result in some limited fuel failure.

** The need to consider complete failure of the shutdown system was removed with the advent of two diverse shutdown systems.

In 1975, designers recognized that for most LOCA they could not show that ECCS would prevent significant fuel failure; on the contrary, safety analyses were predicting extensive fuel failures. This recognition identified, inter alia, the need to review more closely the capabilities and weaknesses of CANDU containment designs. Not surprisingly, for single failures when the containment was assumed to function as designed, the prompt releases of fission products could be shown to be low. However, for a variety of dual failures, i.e. those which consisted of a LOCA plus some impairment of containment functions, the releases of fission products were calculated to be very significant and in some cases in excess of the dual failure reference dose limits. This new information led to the current requirement to analyze in detail the consequences of a wide variety of containment impairments (e.g. deflated airlock seals, open airlock, impairment of dousing, failure of reactor building air coolers) which had previously been passed over or superficially investigated. The new information on ECCS effectiveness also led to investigations of improvements in design to reduce the probability of impairment of containment (e.g. alarms on air supplies to airlock door seals, improved environmental qualification of equipment) and to improve post-LOCA operation (e.g. shielding of equipment, improved filtered air-discharge systems).

2.2 Impact of Commissioning Results

Another factor which has influenced the regulatory requirements for containment (as well as the other special safety systems) has been the results of commissioning tests at plants. A great deal of emphasis is put on commissioning tests to demonstrate, as far as practicable, that all containment systems meet their minimum allowable standards. Clearly, in a 600 MW reactor design it is not practicable to demonstrate by test the dousing rate. Just as clearly, it is practicable and it is a requirement to do the building pressure proof test very late in the commissioning phase after all the containment penetrations have been completed. Similarly it is practicable to test the dousing system in a vacuum building by opening simultaneously all the building pressure relief valves. The requirement for very comprehensive commissioning tests has been supported by unexpected results on many occasions. At Pickering A and Bruce A, for example, the commissioning tests of the dousing systems demonstrated that dousing would not occur under design conditions. (The corrective action at Pickering A was simple - change the operating pressure in two chambers. The corrective action at Bruce A was more difficult - replace major portions of the original dousing pipework to prevent collapse of piping which occurred under design conditions.)

2.3 Impact of Operating Experience

As one would expect, the regulatory requirements for containment have also been affected by operating experience. This aspect will be discussed in sections 3 and 4. Suffice it to say at this point that the requirements reflect the fact that the observed leakage of containment buildings in service has not always been acceptable.

2.4 Selected Topics from Regulatory Document C-7

By and large, Regulatory Document C-7 (Reference 1) merely documents requirements which have evolved over the years and have been generally understood by designers and regulators. There are, however, some requirements which are new or are perceived by some to be new. Foremost among these are:

(a) That all piping which is part of the primary heat transport system (with the exception of boiler tubing) shall be totally within the containment structure. This would preclude, for instance, locating the maintenance cooling system outside containment as was done at Bruce A and B.

(b) That there be sufficient shielding to ensure access to all equipment as may be required following an accident. This requires, inter alia, that the radiation fields be predicted for a wide range of potential accidents.

(c) That any gross breach of the containment envelope can be readily and reliably detected.

(d) That a leakage test at full design pressure be done at least once every six years, or more frequently depending on experience with tests at reduced pressures.

(e) That the boundary of the containment envelope be defined for all conditions which could exist in operation or maintenance of the reactor. This requirement is quite straightforward for a CANDU 600 MW design. It is anything but straightforward when one considers a Bruce reactor during maintenance of a primary heat transport pump, replacement of a liquid zone control compartment, or repairs to a maintenance cooler circuit.

3. FACTORS CONTRIBUTING TO CONFIDENCE

3.1 Equipment Availability

Despite the vast array of penetrations into the containment envelope, good availability can be achieved by careful attention to detail in the design and operation. While the early performance of the Pickering A containment system was not satisfactory, Table 1 illustrates that good availability of containment can be achieved. In fact in 1983 the performance of systems at most plants was very good. With the exception of NPD and Bruce Unit 3 all plants met their availability targets.

Reactor	Actual Unavailability (multiple of target)
NPD	11.9
Douglas Point	0.4
Pickering 1	0
Pickering 2	0
Pickering 3	0
Pickering 4	0
Pickering 5	0
Pickering 6	0
Bruce 1	0.3
Bruce 2	0.3
Bruce 3	52*
Bruce 4	0.3
Gentilly-2	0
Point Lepreau	0

TABLE 1
Containment Unavailability
All Reactors - 1983

* Activity monitors were isolated during a reactor outage.

In addition to being able to demonstrate by test the availability of active components, plant operators also have the capability to detect the presence of an opening in the containment boundary without the need to shut a reactor down. This can be accomplished in one of two ways:

i) Closing all ventilation dampers, noting the rate at which building pressure rises due to instrument air in-flow and comparing this rate with that established for a "leak-tight" building. Pressure is allowed to rise only to 1 kPa(g) (four inches of water).

ii) Reducing reactor pressure to 7 to 30 kPa (1 to 4 psig) subatmospheric and observing the rate of pressure rise.

While these methods have obvious limitations they can be very effective in detecting holes in the containment envelope (see results for Pickering in Table 2). As such they are a useful supplement to (but not a replacement for) routine positive pressure leakage tests at the design pressure or at a somewhat reduced pressure.

Reactor Building	3rd Quarter 1983 (mm^2)	4th Quarter 1983 (mm^2)
1	0	1030
2	1030	1660
3	380	1290
4	1700	570
5	260	10
6	-	360

TABLE 2

Containment Hole Size -
Estimated from On-Power Leak Test Target = 2400 mm^2

3.2 Radiation Dose to Operations Personnel

A unique, as well as controversial, aspect of the Bruce and Darlington containment design is the number of nuclear system components which are outside the containment envelope (e.g. boilers, primary system pump sets, drives for reactivity devices, moderator heat exchangers). This feature was adopted by Ontario Hydro to improve on-power maintainability of equipment as well as to reduce radiation dose to personnel. The success in this latter area can be seen by comparing the absorbed radiation dose at Pickering A and Bruce A - see Table 3. The comparison given in Table 3 is over-simplified in that corrections should be made to account for a number of operating differences between the two plants (e.g. heavy water with significant tritium concentration was used for commissioning of Pickering A, pressure tube problems at Pickering A accounted for some of the radiation dose to personnel in 1974, 1975, 1976 and 1983). Nevertheless, even without a rigourous treatment of the dose statistics it is obvious that the Bruce A type of design results in much lower radiation doses to the operating staff.

YEAR	PICKERING A (man-rem)	BRUCE A (man-rem)
1971	150	-
1972	992	-
1973	899	-
1974	1613	-
1975	1627	-
1976	1740	10
1977	1143	104
1978	822	384
1979	1129	498
1980	893	410
1981	778	472
1982	817	369
1983	1192	421

TABLE 3

Total Whole Body Dose to All Staff

4. PROBLEM AREAS

4.1 In-Service Leakage Testing

A major uncertainty about CANDU containment system effectiveness stems from the absence of a comprehensive program for doing in-service leak tests. A comprehensive program would consist of

i) periodic on-power leak tests of major containment penetrations.

ii) periodic on-power leak tests of the entire containment envelope.

iii) periodic leak tests of the entire containment envelope at its design pressure or alternatively a combination of reduced pressure and design pressure tests.

To date the three Canadian utilities have implemented item i); that is periodic leak tests of airlocks and ventilation dampers. Even in this area, however, the tests are sometimes token in nature when compared with what could be done readily. For instance, at Pickering, the airlocks are tested at a pressure of 1 kPa(g) (4 inches of water) when the entire building is tested. While that pressure is a reasonable one for an on-power test of a reactor building, it is obvious that a more rigourous test of airlocks could be done readily without interfering with reactor operation.

Periodic on-power leak tests of the entire containment envelope have been demonstrated by Ontario Hydro as a relatively simple and yet sensitive test for detecting existing holes in containment. Such tests are done routinely at the Pickering and Bruce stations. They are not being done yet at Gentilly-2 or Point Lepreau reactors.

The history of in-service testing of containment at its design pressure is not an encouraging one. Only two reactor buildings in Canada have been retested in service at their design pressures. Both failed the tests. The first to be retested in service was the Douglas Point reactor building. The results of all the tests are summarized in Table 4. While this building

failed to achieve its specified leak rate on several occasions, it is evident that by retesting and repairs the leak rate could be kept in an acceptable range.

Gentilly-1 was the only other plant where the reactor building was tested in-service at its design pressure. After seven years of "service", a test was attempted in 1977. In the first attempt leakage started around an airlock at a pressure of about 35 kPa(g) and another large leak started in a similar location at 90 kPa(g) (Design pressure 120 kPa(g)).

When Pickering A was being constructed and when it was licensed for operation Ontario Hydro indicated that it would do periodic leak tests of reactor buildings at 14 kPa(g) (1/3 the design pressure). One successful in-service test was done on Unit 3 in 1975 but no further tests were done until 1983 when the AECB ordered such tests to be done. Two tests, with satisfactory results, have now been completed with a third to be done in September, 1984.

The reasons why plant owners wish to conduct only necessary tests are clear. Shutdown of a single unit to test a reactor building is expensive, results in additional radiation dose to workers and, in the case of the 600 MW designs, involves the hazard of working in a high pressure environment. Shutdown of four or eight reactors to allow testing or inspection of a common vacuum building is indeed a major undertaking. Furthermore, in multi-unit containment designs, unless one is prepared to test all the units at the same time, there is the additional difficulty of isolating one unit from another. These costs and difficulties are real but they are costs which were foreseen when a commitment was made to build a plant and they must be incurred if a utility wishes to assure people that it takes all practicable measures to demonstrate the effectiveness of special safety systems.

DATE	Leak Rate - % of Reactor Building Volume per Hour at Design Pressure
Target	0.1
1965	0.07
1968 July	0.40*
1968 Oct	0.12*
1969	0.33*
1970	0.15*
1973	0.06*
1980 (5 tests)	0.14 to 0.36
1981	0.18
1982	0.18
1983	0.25

TABLE 4
Douglas Point - In-Service Leak Testing of Reactor Building

* extrapolated values - test performed at 1/3 of the design pressure.

4.2 Reliability of Containment Envelope

Section 3.1 included information on the reliability of components in the containment system. Notwithstanding the good performance in 1983, experience shows us that there are rare events which can have significant impact on containment effectiveness and are difficult to detect through routine testing.

One example of such a rare event was a 130cm^2 hole (20 in^2) in the Pickering Unit 2 reactor building at a pipe penetration. It is not known how long this fault existed but circumstantial evidence indicates somewhere between 6 months and 3 years. (With current practice at Pickering such a fault would not exist for more than 3 months, the period between routine on-power leak tests). The problems discovered by leak tests at Gentilly-1 (as discussed in Section 4.1) are yet another manner in which containment envelope impairments can go undetected for significant periods.

Another example of a rare event occurred at Douglas Point in 1980. In that event, all the activated charcoal had been removed from the reactor building ventilation exhaust filter while the reactor was shut down. It was not replaced and the reactor operated for 6 months with this significant impairment.

..her rare events include:

i) One of the two upper chambers of the vacuum building at Pickering A was at atmospheric pressure (rather than 15 kPa) for a period of months because the instrumentation was monitoring conditions at the vacuum pump suction rather than in the chamber.

ii) Failure of 11 of the 16 large reactor-building pressure relief valves at Bruce A in such a fashion as to preclude opening because the design had not taken into account test conditions which the valves would encounter.

iii) Failure by corrosion of wire rope cables which hold in place the Bruce A vacuum building roof seal.

Experience shows us that not only must we continue to check routinely the proper functioning of all the active components in the containment system but in addition we must take all reasonable precautions to avoid failures which are not readily detectable by test. "All reasonable precautions" include attention to detail in design and operation. For example, the safety evaluation and operating experience show that an important failure mode is inadvertent deflation of airlock door seals. An extremely simple design change could be made at existing stations, (and in some airlocks of one station this change has been made), which would reduce the resultant flow path to atmosphere by one or two orders of magnitude. Another reasonable precaution for operators to take would be to check routinely the efficacy of the charcoal beds in the post-LOCA containment filters. Even this is not done at all stations.

4.3 Post-Accident Air In-Leakage

With careful attention to design, operating procedures, routine testing of active components and periodic in-service leak tests it is practicable to meet the targets for both effectiveness and reliability of containment. However, in the design of CANDU plants we have a feature which tends to work

against all the care and attention that owners have put into containment design. This feature is the continued flow of instrument and service air into containment following any accident. After an accident which releases large quantities of radioactive material into containment, the containment would be sealed through automatic operation of the ventilation dampers. The pressure in the reactor building would rise due to the continued inflow of service and instrument air and this would result in pressure-assisted leakage of fission products from containment.

The problem would be most immediate in the 600 MW containment design. Designers have taken some account of this and in the 600 MW design it is possible to isolate manually all air flow into containment at the expense of the operability of some equipment. The Pickering and Bruce containment systems with their vacuum buildings are able to defer pressure-assisted releases by periods ranging from one or two days (unimpaired containment) down to an hour (with impaired containment envelope). However, after that time the continued inflow of instrument air would result in pressureassisted release of fission products via leakage pathways. Venting via the filtered air-discharge systems would be required to reduce releases of radioiodines. (The safety analysis does show that even with required venting of containment the relevant single and dual failure reference dose limits would be met at the Bruce and Pickering sites).

Clearly, it is not appropriate to design an effective and reliable containment only to include design features which will require operators to vent radioactive material to the atmosphere soon after an accident. The AECB has requested designers to investigate this question at Pickering and Bruce and expects that improvements will be proposed.

5. FUTURE CONSIDERATIONS

5.1 Hydrogen Research

In the last several years there has been a significant amount of research done at Whiteshell to study hydrogen generation and behaviour under accident conditions. One objective of this work is to determine whether any design or operating changes are necessary in existing CANDU plants. Partly as an outcome of this work and partly as a prudent measure Ontario Hydro has decided to include hydrogen igniters in the Bruce A, Bruce B, Darlington, and Pickering B reactor buildings. These ignitors are intended to ensure that, should large quantities of hydrogen be produced, it will burn while the concentrations are still relatively low and thereby prevent deflagration or detonation. Results from the research program are expected to answer the question regarding need for igniters or other design changes at all plants.

5.2 Fission Product Behaviour

Almost two decades ago, work was done at Oak Ridge National Laboratory which indicated that overly pessimistic assumptions were probably being used in safety analyses. It was recognized then that further research work was necessary to study fission product behaviour under accident conditions particularly with respect to behaviour of iodine within containment. In Canada this confirmatory work was undertaken about four years ago at the Whiteshell Nuclear Research Establishment. Continuation of such work is crucial if we are to have a proper understanding of accident scenarios.

5.3 In-Service Leakage Testing

In the past, in-service leakage tests at significant positive pressure have been relatively rare events. The AECB has partially addressed this shortcoming by including specific requirements in the operating licences for plants. We expect to see the development of more comprehensive in-service leak rate testing programs and trust that these will be successful in demonstrating the capability of CANDU containment structures.

6. CLOSING REMARKS

CANDU containment systems are capable of limiting releases of radioactive material under a wide range of postulated accidents. Improvements in some areas are being made while other improvements are still under investigation. High reliability and effectiveness of containment can only be achieved through the efforts of designers and operators who are committed to the principle of defence-in-depth.

REFERENCES

(1) Atomic Energy Control Board - Consultative Document C-7 Rev. 1 - "Requirements for Containment Systems for CANDU Nuclear Power Plants" May 21, 1982.

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